Historical Perspective MSRE Safety Basis Authorization

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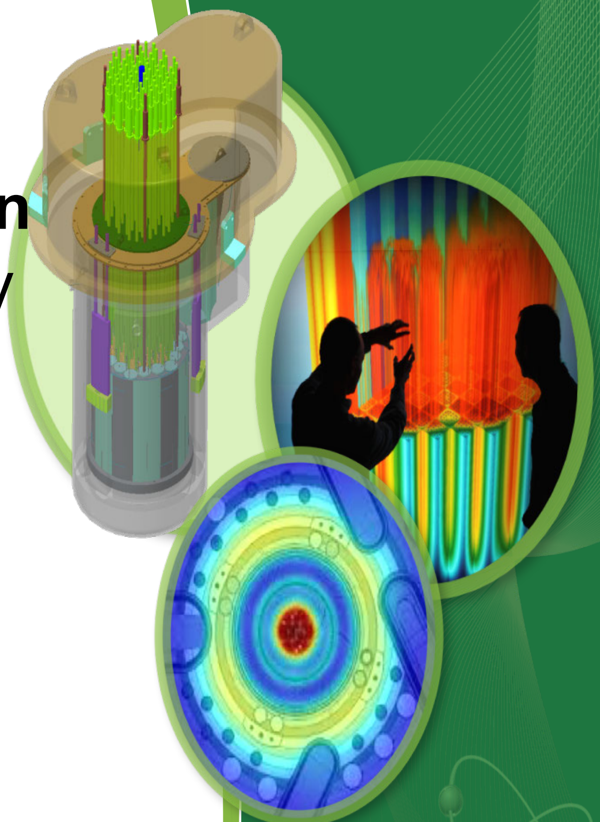

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Topics Discussed

- History of Licensing Non-LWRs
- AEC process for licensing an experimental reactor
- MSRE licensing results

Evaluation and Licensing of Non-Light Water Reactors (LWRs) Dates Back to 1950s

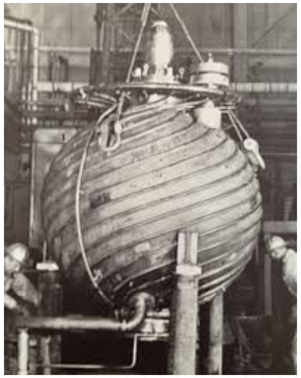
- The Atomic Energy Commission (AEC), the Advisory Committee on Reactor Safeguards, and after 1974, the US Nuclear Regulatory Commission have a long history of evaluating and licensing of non-LWRs starting with Experimental Breeder Reactor (EBR I) in 1951—credited with the first significant power generation
- Nine gas-cooled reactor designs were evaluated or reviewed, not counting the mHTGR or Next Generation Nuclear Plant (NGNP)
- Nine sodium-cooled reactor designs not counting PRISM/SAFR were evaluated or reviewed
- Numerous one of a kind research and test reactors were also evaluated or reviewed
  - Test reactors (materials testing)
  - Isotope production

AEC Licensed Several Other non-LWRs for Experimental Purposes

- NaK cooled
- Heavy water moderated
- Air cooled graphite moderated
- Sodium cooled graphite moderated
- Water cooled graphite moderated
- Organically moderated and cooled
- Liquid fueled systems
  - Molten Salt
  - Aqueous Homogeneous
- Space Reactors
  - Nuclear Rockets
  - Space Power Reactors








Early Reviews Were Done Without the Availability of the Regulatory Guidance and Structure Established for Current LWRs

- Early reviews were customized and based on engineering experience and judgement of participating individuals
- As LWR reviews became more numerous ~1960, reviews also became more objective and regulatory guidance was developed, which provided structure for both the applicant and the regulator.
- For more current non-LWRs (FFTF) explicit use was made of the LWR guidance where applicable, the practice continues today
- Reactors built under the Cooperative Power Reactor Demonstration Program were licensed under the Part 104 licensing process (research and testing reactors)
  - Congressionally mandated joint cost/risk sharing program between the AEC and private industry to promote commercialization of nuclear power

Construction Permits and Operating Licenses were Granted for Two Commercial Gas-cooled Power Reactors

- Peach Bottom 1
  - 40 MWe Philadelphia Electric Co. (Cooperative Power Reactor Demonstration)
  - Construction Permit (CP) 1962, Operating License (OL) 1967, shutdown 1974
  - AEC review (104 license)
- Fort St Vrain
  - 350 MWe Public Service Company of Colorado
  - CP 1968, OL 1973, shutdown 1989
  - AEC review

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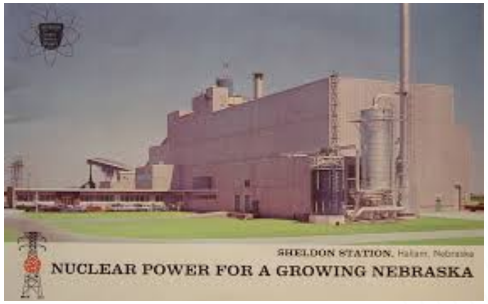
Construction Permits and Operating Licenses were Granted for Two Sodium-Cooled Reactors

- Hallam Nuclear Generating Station (sodium-cooled graphite moderated reactor)
  - 75 MWe Consumers Public Power District (Cooperative Power Reactor Demonstration)
  - CP 1959, OL 1963, shutdown 1964
  - AEC review (104 license)
- Fermi 1 Nuclear Power Plant (metal fueled fast breeder reactor)
  - 69 MWe Power Reactor Development Co. (Cooperative Power Reactor Demonstration)
  - CP 1956, OL 1963, shutdown 1972
  - AEC review (104 license)



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Construction Permit and Operating License was Granted for a One-of-a-Kind Reactor

- Piqua Nuclear Power Facility (organically-cooled and moderated reactor)
  - 12.5 MWe Piqua Municipal Utilities, Piqua, OH (Cooperative Power Reactor Demonstration)
  - CP 1957, OL 1963, shutdown 1966
  - AEC review (104 license)

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DOE-Regulated Fast Flux Test Facility (FFTF) Used NRC to Perform the Technical Review

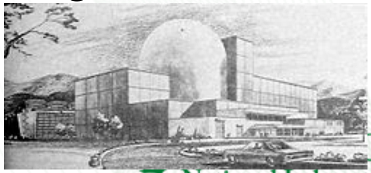
- FFTF was a sodium-cooled test reactor built on the DOE Hanford site
- AEC, ERDA, and later DOE regulated facility
  - Used NRC and ACRS to re-evaluate the design prior to operation
  - PSAR submitted to AEC, September 1970
  - NRC formed in 1974
- ERDA/DOE requested NRC/ACRS review of the PSAR
  - Did not require an NRC License to operate
  - But did request an NRC Technical Review
    - To be sure this facility would meet the strictest, independent regulations
    - To bring the NRC up to speed on licensing sodium-cooled fast reactors

  NOTE: DOE expected fast reactors to be built soon in rather large numbers, so they desired an in-depth FFTF regulatory review to "jump start" the licensing review process for LMFBRs

  - NRC Safety Evaluation Report issued Aug.1978 (supplement 1979)

Clinch River Breeder Reactor (CRBR) Licensed by NRC (Commercial) Using 10CFR50

- CRBR was a joint demonstration project between DOE and private industry (TVA and Commonwealth Edison)
- 380 MWe sodium-cooled fast breeder reactor was to be built at a TVA site near Oak Ridge, TN
- Intended to be licensed as a commercial reactor under NRC
- Followed the existing LWR process
- Required exemptions/exceptions, and modifications to the existing LWR regulatory criteria (e.g., General Design Criteria were revised to reflect unique LMFBR aspects of the design- Issued as ANS 54.1)
- Two phases of licensing
  - Licensing began 1974, NRC work stopped by President Carter in 1977
  - September 1981 licensing renewed, stopped in 1983 by Congress
  - Received a Limited Work Authorization (LWA) in 1983
  - ASLB positive finding but no construction permit



The Licensing of non-LWRs is Not New

- For 20 years the AEC licensed a wide variety of non-LWRs mostly using customized reviews based on engineering experience and judgement
- Starting in the 60s when LWRs began to be the reactor of choice, the reviews became more objective evolving into what is the current set of LWR focused regulations

Non-LWRs since the late 1960s timeframe have been licensed using the same process as the LWRs but using exceptions and exemptions where LWR requirements are not adequate or do not apply

Until Regulations were Issued in the late 1960s Licensing for Both Experimental Reactors and Commercial Reactors Began with a Preliminary Hazards Analysis (PHA)

- Formed the basis for the safety analysis
- Reviewed by the AEC Regulatory Division (Division of Reactor Licensing) if commercial
- Experimental and Test Reactors followed a less prescriptive pathway
- PHA and Safety Analysis Report (SAR) was reviewed by the Advisory Committee for Reactor Safeguards (ACRS)
- In the SAR, the applicant proposed a Postulated Maximum Credible Accident
  - Bounded all other accident consequences - used to determine the offsite consequences
  - Credible (used to be hypothetical) meaning within reason (but not necessarily mechanistic)
  - Set up the process for a LWR LBLOCA (used as the LWR maximum credible accident)
  - Later became the design basis accident (accidents that form the design basis for engineered safety features)

Approach Taken by Experimental, Research and Test Reactors at ORNL

- Reactor design was reviewed by an internal independent group of experts within ORNL reporting to the Laboratory Director (9-15 experts from various disciplines)
  - Reactor Operations Review Committee (RORC)
  - Reviews began with conceptual design and continued through construction and operation
  - PHA and Safety Analysis documentation was prepared by ORNL project team and reviewed by RORC
- Results of the internal review of the PHA and SAR along with the actual documents were presented to the AEC Oak Ridge Operations (ORO) office for review and comment
  - Usually set up an independent review committee of outside consultants
  - ORO Review results were sent to the program office at AEC headquarters (for MSRE the Reactor Technology Development Office was the program office)

AEC Headquarters Was the Authorizing Organization based on the Atomic Energy Act

- The PHA and SAR were reviewed by the program office and the ACRS (the internal and ORO reviews substituted for the Division of Reactor Licensing Review)
- If approved these groups made a recommendation to the Atomic Energy Commission (5 politically appointed members)
- AEC then granted a construction permit for the reactor
- The same process was repeated for the operating license.
  - An operational readiness review was conducted by ORO and the project office at AEC headquarters prior to startup.

MSRE PHA and SAR Were Based on Previous Experience with Fluid Fueled Reactors

- Aircraft Nuclear Reactor Experiment
- Two Aqueous Homogeneous Reactor Experiments (HRE 1 and 2)
- MSRE PHA was first published in 1960 and reissued in 1962 "Molten Salt Reactor Experiment Preliminary Hazards Report", ORNL CF-61-2-46 Addendum No. 2 (May 8, 1962**)**.
- Supporting safety analyses were published in Safety Calculations for MSRE ORNL- TM 251 (1962)

PHA Was a Barrier Analysis

- Each area of the MSRE that contained fuel or fission products was surrounded by at least 2 barriers
- Events were identified which could partially damage a single barrier- no release if second barrier remained undamaged
- Events that might damage two barriers were those that were considered to contribute to off-site consequences. Today these might be considered Beyond Design Basis Events
- Identified events were then analyzed as part of the safety analysis to provide detailed information on fuel/graphite and vessel temperature, power, and pressure

Six Reactivity Events Were Analyzed in the SAR using analog computers and a digital computer code MURGATROYD, later ZORCH

- Fuel Pump Failure
- Cold slug accident
- Filling Accident
- Loss of graphite from the core (filling empty space with fuel)
- Fuel addition (precipitated fuel circulating in core or non mixed fuel- lumps circulating in core)
- Uncontrolled control rod withdrawal
- Also looked at various ramp/step reactivity additions

Results Indicated that MSRE Accidents were Benign

- None of the accidents led to catastrophic failure of the reactor
- Some internal damage to the reactor from high temperatures could result from three events: extreme cold slug accidents, premature criticality during refueling, and uncontrolled withdrawal of control rods.
- These extreme events could only result from compound failure of protective devices, in each case there existed effective corrective actions independent of the primary protection so damage was unlikely
- Arbitrary ramp and step additions - damaging pressure would occur from a step of 1% Δ k/k ($2.8) or greater, $\beta_{eff=0.0035}$

Final Safety Analysis of MSRE (ORNL- TM- 732)

- In addition to reactivity events the SAR examined
  - Loss of Flow
  - Loss of Heat Sink
  - Decay Heat Removal
  - Criticality in Drain Tanks
  - Freeze valve and flange failures
  - Excessive wall temperatures
  - Corrosion
  - Salt spillage
  - Be release from a leak
- Most probable accident- small leak into secondary container
  - Radiation monitors would alarm and shut down reactor
  - Airborne activity pumped from secondary containment through clean up system and filters released up the stack did not exceed maximum permissible dose on-site

Final Safety Analysis of MSRE

- Maximum Credible Accident
  - Break in drain line (1 ½ inches) 10,000 lbs. salt released to secondary containment
  - Or Break in 5 inch fuel line (4000 lbs. salt released)
  - Assumed both total 10,000 lbs. (4000 from fuel and 6000 from drain line in 280 sec.
  - Simultaneous spillage of water into secondary containment to maximize steam pressure
  - 110 psig (no venting)
  - Rupture disk opens at 20 PSIG to vapor condensing system
  - Maximum pressure in secondary containment is 39 psig (no rupture)
  - 1% leakage at 39 psig
  - Dose offsite (3000 m) is 6 rem from Iodine under worst meteorological conditions
    - 10% iodine, 10% solids,100% nobles

SAR Was Originally Part of a Series of Design and Operations Reports Submitted to the AEC for Review- Provides Safety Basis for the License

- SAR
- Reactor Design description
- I&C description
- Nuclear Design
- Chemistry and Materials
- Operating Limits
- Fuel handling and processing plant
- Operating Procedures
- Safety features and emergency plans
- Maintenance equipment and procedures
- Test Program
- Drawings and supporting design information